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REACTOR VITAL EQUIPMENT DETERMINATION TECHNIQUES

by

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ABSTRACT

The Reactor Vital Equipment Determination Techniques program at the Los Alamos National Laboratory is discussed. The purpose of the program is to provide the Nuclear Regulatory Commission (NRC) with technical support in identifying vital areas at nuclear power plants using a fault-tree technique. A reexamination of some system modeling assumptions is being performed for the Vital Area Analysis Program. A short description of the vital area analysis and supporting research on modeling assumptions is presented. Perceptions of program modifications based on the research are outlined, and the status of high-priority research topics is discussed.

I. INTRODUCTION

In 1978, NRC requested technical assistance from Los Alamos in determining vital areas based on 10 CFR 100 release criteria for all commercial nuclear power plants in the United States. A systematic analysis based on a fault-tree method had been developed at Sandia National Laboratories, Albuquerque, in the early 1970's for the NRC Office of Research (RES). Since 1978, Los Alamos has applied the method to specific plants for the Office of Nuclear Reactor Regulation (NRR) and, more recently, for the Office of Nuclear Material Safety and Safeguards (NMSS). This technique has proven to be an excellent tool for performing detailed and systematic analyses of complex plants.

The vital area fault-tree methodology uses the SETS computer code to determine cut sets and provide usable information on vital areas. The construcplants. Because NRC requirements have changed with respect to the VAA, some analyses performed before 1980 are being redone to provide information not required when the original analyses were completed. In addition, plants receiving operating licenses are analyzed as they come on-line. Consistent analyses of all plants will be available within a few years. These analyses are being used by NMSS in its Regulatory Effectiveness Review (RER) program. The results also have proven useful to at least one utility when considering the reconfiguration of its vital areas.

II. VITAL AREA DETERMINATION RESEARCH PROJECT

In order to construct a fault-tree model of a system, analysts must make numerous decisions concerning success criteria, consequences of events, and system dynamic response. This invariably leads to some decisions being based on incomplete knowledge. Complete understanding is replaced by reasonable assumptions in the decision-making process. Invariably, new information eventually becomes available that affects the decisions made in constructing the fault trees. This necessitates a reexamination of the fault-tree assumptions and can change the results based on these models.

Some vital area analysis assumptions currently are being reevaluated. The assumptions to be examined were identified by an independant NRC working group formed for the review of the VAA. The working group identified 11 topics for reexamination. These topics are listed in Table I. The first phase of the vital area determination research was to survey existing literature for information related to these topics. Ongoing research projects promising information of interest also were identifed. Computerized data bases relating to nuclear safety were searched, and over 9500 citations were noted. Of these, 114 were deemed sufficiently interesting to warrant reading. Finally, 61 reports were abstracted as directly applicable to VAA needs.

The 11 topics were categorized into three groups based on the results of the survey.

(A) Topics for which sufficient information already exists to resolve any uncertainties.

(B) Topics for which ongoing research appears adequate for resolution of uncertainties.

(C) Topics for which there is inadequate information or ongoing research to address uncertainties from the VAA perspective.

TABLE I

TOPICS FOR REACTOR VITAL EQUIPMENT DETERMINATION
TECHNIQUES RESEARCH STUDY

1. Identifying individual safety-related cables in cable trays.

2. Disabling complete cable trays.

3. Disabling systems needed during shutdown or refueling conditions.

4. Disabling sensor systems, instrumentation, and nonsafety-related control systems.

5. Treating spatially extended systems and components [that is, piping; electrical distribution; and heating, ventilating, and air conditioning (HVAC) systems].

6. Scenarios involving air systems.

7. Disabling electrical equipment by grounding or lifting of grounds.

8. Relating best-estimate analyses of plant responses to system failures to the corresponding Final Safety Analysis Report (FSAR) analysis.

9. The effective inclusion of random events, such as anticipated transients, in fault-tree methodologies.

10. Possible system failures after which stable hot shutdown cannot be maintained indefinitely.

11. Considering the use of nonsafety-related equipment, unanalyzed procedures, or operator ingenuity to recover from system failures.

The categorization of the topics is presented in Table II. Some topics are very broad in scope and include several subtopics (for example, use of best-estimate analyses.) In these cases, different subtopics fall into different categories.

The next step in the analysis procedure was to set priorities on research topics. These priorities were based on ease of resolution and effect on the fault-trees. The topics fell into four groupings.

(1) Topics resolvable with limited effort and affecting the fault tree significantly.

(2) Topics requiring considerable work and having significant effect.

(3) Topics with many subtopics requiring great effort and having potentially major effect.

(4) Topics that are not considered to have major effect because they do not represent direct, high-probability paths to an event of concern in VAA (Part 100 release).

This priority grouping is shown in Table III and will guide future research in this project.

TABLE II

CATEGORIZATION OF TOPICS

Topic (See Table I.)	Category (See text.)
1	B
2	B
3	C
4	B, C
5	C
6	C
7	C
8	A, B, C
9	C
10	B, C
11	B, C

TABLE III

PRIORITY GROUPINGS

Priority Groups	Topics
I	1, 2, 3, 10 (See Table I)
II	9, 4
III	8, 11
IV	5, 6, 7

III. CURRENT WORK

No topics were resolved totally based on current literature; therefore, all topics require monitoring of ongoing projects or some new research. However, the topics in priority category I were resolvable within a short time, and work has been completed on topic 1 and is underway on topics 2, 3, 4, and 10. These topics are addressed individually.

A. Viability of Identification of Individual Safety-Related Cables (Topic 1)

The potential for identifying individual safety-related cables in cable trays was examined by contacting plant engineers, electrical maintenance personnel, electrical contractors, and architect-engineers. The general consensus was that such a procedure is nearly impossible and requires extraordinary knowledge and determination under the best circumstances. Two plants were identified by an architect-engineer as having marked cables. These plants were contacted and maintenance personnel were questioned. The personnel acknowledged that safety-related cables originally were marked; however, newer and replacement cables were unmarked, and many original marks now are obliterated. They did not consider the identification and subsequent damage of individual cables to be a feasible or a direct path for causing plant damage. Therefore, the current fault-tree modeling in which individual cables are not treated as targets is felt to reflect realistic plant conditions.

B. Maintaining Hot Shutdown Conditions (Topic 10)

One area of concern in maintaining stable hot shutdown conditions for pressurized water reactors (PWRs) is loss of reactor coolant pump (RCP) seal integrity during a station blackout. In current fault-tree modeling for a station blackout, if a steam-driven auxiliary feedwater pump and steam generators are maintained, primary system capability is preserved using natural circulation. If an RCP seal failure were induced by a loss of all RCP seal cooling water, hot shutdown would be untenable unless electrical power were provided to some primary makeup pumps. This would require protecting onsite power sources and associated essential auxiliaries as well as component cooling water.

The severity of RCP seal leak was scoped with a TRAC-PF1 calculation of a large Westinghouse PWR. This calculation assumed a steady-state natural circulation cooling mode induced by auxiliary feedwater cooling. It was assumed that three RCP seals failed 30 min after shutdown, causing three 400-gal/min seal leaks. No primary makeup was modeled.

The calculations indicated a primary depressurization to saturated conditions in 20 min. Natural circulation was suppressed at 50 min. The core continued to cool by an efficient reflux cooling mode that rejected heat through steam leakage out the pump seals and condensation on cool reactor internals with drainage back into the core. This is similar to other TRAC small-break calculations. The calculation was terminated at about 4 h when core cooling was finally lost. Recent work suggests that the time to RCP failure may occur long after loss of cooling water, and leak rates may be well below the assumed 1200 gal/min. This would indicate that inducing RCP seal leakage is not a direct, high-success-probability scenario for core damage because of the long times involved. More investigation of seal failure sequences and modes is required before final conclusions are drawn.

C. Reactor Vulnerability While Shut Down or Refueling (Topic 3)

In current fault trees, it is assumed that the reactor operating at power is in its most vulnerable configuration, so vital areas identified based on this analysis include as a subset vital areas under other reactor conditions. A careful analysis of reactor plants in shutdown has been undertaken, and some new scenarios have been identified. Work is underway to construct fault-trees modified for shutdown conditions for a representative PWR and a boiling water reactor (BWR) to determine the effect of these conditions on the VAA. If these model trees indicate a need, the effect on specific plants will be addressed.

D. Feasibility of Core Damage Because of Destruction of Cable Trays (Topic 2)

Currently, only areas where all cable trays are located are considered in the VAA (for example, cable spreading rooms). The accuracy of the current fault-tree model is being investigated by first identifying scenarios requiring destruction of entire cable trays and the probability of success for various modes of destruction. This work is just beginning, and no definitive results are available.

E. Best-Estimate Containment Strength (Topic 8)

In the VAA, the containment is considered breached if core melt occurs. However, the question of containment integrity with no core melt is of some concern. If a PWR containment is pressurized by a loss-of-coolant accident (LOCA), conservative assumptions on containment strength and pressure rise lead some utilities to require containment cooling systems to prevent possible containment breach. If the containment is breached, radioactive steam is released and RHR pumps may cavitate in the recirculation mode, threatening loss of long-term cooling. Thus, containment cooling systems appear in some PWR fault trees.

A simple, conservative calculation indicated that a 10 CFR Part 100 release is not possible for a plant operating with design coolant activity without some core damage. This is in accordance with current vital area assumptions. Past research has indicated containment strengths far in excess of FSAR values, and current work in the severe accident area promises to provide information on penetration strength. When this information becomes available, a recommendation on containment modeling will be made. It should be noted that no change to Type I vital areas at any plant is expected. The possible results of this work would be changes in Type II areas at a few plants where a LOCA can be caused outside Type I areas.

IV. CONCLUSIONS AND FUTURE WORK

Future work in this program includes tackling the more extensive research projects and following ongoing programs that seem to offer guidance on VAA topics. Some or considerable extension of some ongoing research may be required to address specific VAA concerns. Thus, the overall outlook for this project is gradual resolution of outstanding questions in VAA through a combination of monitoring and adapting outside research and performing some well-focused internal work to extend and augment other efforts.

We feel that this work provides a format for probing areas of reactor safety that usually are not explored because the VAA is not limited to single-failure criteria. Thus, complex interactions between several elements of the reactor system are explored. As this program progresses, we feel that the results will be of interest not only to safeguards concerns but in the safety area as well.